Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/629
ITALY
February 1964
Submitted in: ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

STABILITY ANALYSIS AND DYNAMIC PERFORMANCE OF A CHEMICAL SYSTEM OPERATED FOR THE AUTOMATIC FINE CONTROL OF A NUCLEAR REACTOR

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An automatic fine control system has been designed and satisfactorily operated, where the regulating rod and velocity servo are replaced by a boric acid solution circulating in a tube near the reactor core.

Any reactivity variation up to 200 pcm is obtained by injecting pure water or a concentrated boric acid solution into the main hydraulic circuit. This circuit is of the closed-loop type and an expansion tank with an overflow is provided for removing the excess flow during reactivity transients.

A control of the reactivity by means of soluble poison instead of conventional solid-state absorbers offers many advantages, in both power and research reactors. Nevertheless, some additional problems must be dealt with, involving the chemistry of acid solutions, the behaviour of materials and the system dynamics.

Dynamic stability and performance analysis are two of the main factors in the design of a chemical system suited to the fine control of nuclear reactors

This paper illustrates in detail the study performed on a linearized model of the actual system by using the conventional graphic and analytical method of the feedback system theory.

Calculations were made with the aid of an IBM-704 digital computer and the Nyquist diagram was obtained for the open-loop transfer function of the whole system.

Some typical features of a closed-loop circulating soluble poison control system are:

1. A strongly variable gain of the system in function of the value of operating steady-state concentration. This is due mainly to the different effects obtained by injecting an equal amount of water or concentrated boric acid solution; obviously a higher pure-water effect is obtained on increasing the solution concentration - and the opposite happens in the case of a boric acid injection.

   Another effect results from the non-linear characteristic of the reactivity versus concentration diagram for a U-shaped tube employed in the in-pile section because of the self-absorption phenomena occurring at elevated values of boric acid concentration.

2. The time delays caused by the flowing of the solution have two major effects which strongly influence the dynamic behaviour of the system. Firstly, the delay between the injection point and the reactor compromises the system stability being equivalent to a long-time constant; secondly, a tendency towards self-sustaining oscillations is observed at $\frac{1}{T}$ frequencies, where T is the total transit time.

For obtaining a control loop with a dynamic performance similar to that of a conventional mechanical rod drive and velocity servo-system, proportional, integral and derivative controllers were found to be inadequate and a more elaborate network was designed.

The validity of the linear model is then checked by simulating the whole system on an analogue computer; some non-linear effects, such as the valve opening saturation, are found to be important in reactor transients involving heavy reactivity insertions, and more precise controller parameter values are obtained.

The control system was successfully tested and operated on the AVOGADRO reactor. Power transients typical of a research reactor routine operation were performed, as for instance reactivity insertions and variations in power demand and a precision higher than 1% was obtained in the neutron flux level regulation.

The transfer function of the system was also measured and the experimental points were in excellent agreement with the Nyquist diagram.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/630
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

ON SOME PHYSICAL PROBLEMS OF THE URANIUM BURN-UP IN HEAVY WATER REACTOR

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The calculation of the uranium burn-up in heavy water heterogeneous reactors consists of the solution of fuel balance equations system, of the fission products influence calculation and of the long-term kinetics calculation based on the two-group theory. The dependence of the neutrons multiplication coefficient on irradiation (integral flux) in the point of homogenizable medium (so-called macro-burn-up) is numerically determined with the help of the BODKUR 2 programme for the automatic computer Ural 2.

The actual burn-up of uranium (micro-burn-up) in a reactor is given by the way of fuel discharge in the reactor. For the case of continuous discharge of the whole fuel elements, when there are elements in the most varied stages of burn-up in the reactor, the micro-burn-up is a simple function of the macro-burn-up and its calculation is included in the BODKUR 2 programme. The results of calculations show that in the operating extension of the burn-up of natural uranium in a heavy water reactor the dependence of reactivity on irradiation can be with satisfactory accuracy approximated by the quadratic curve and thus the further analysis is substantially facilitated.

From the point of view of the original operation of a reactor one question is important: that of the transition of the reactor from the original state - with all elements being fresh - to the equilibrious state with continuous discharge, when there are elements of different burn-up in the reactor, ranging from entirely fresh to the completely burnt-up ones. It appears that when choosing the actual way of operation in the transient state the following requirements are necessary to be kept in view:

1. The minimum length of the transient state of the reactor both from the point of view of reactivity and replacement of elements;
2. The maximum necessary radial stability of the element;
3. The maximum attainable burn-up of fuel;
4. The maximum necessary velocity of the elements replacement.

The circle of suitable ways of transition is limited by these requirements. In the paper a few transition ways are discussed and it is shown that if the maximum permissible irradiation of elements in the equilibrious state is decisive, then, at any kind of transition from the original to the equilibrious state, the burn-up of one half of the average equilibrious irradiation of one fuel charge always will be lost.

In addition to the discharge of the whole elements there are, of course, other ways of fuel shuffling in a reactor. One of them is e.g. the continuous shuffling of fuel in the direction of the reactor axis. In the paper is investigated the axial flux distribution and the attainable burn-up for the fuel shuffling in one direction, the shuffling in opposite direction (one half of fuel moves in one direction and the other in opposite direction) and the regressive shuffling (the fuel returns back after reaching the second base of reactor).

During these studies the problem of the accuracy of thermal parameters of a reactor occurred. An attempt was made - that of application of thermodynamics of irreversible processes on the calculation of effective temperature of neutron gas. The received results for the homogeneous and heterogeneous medium are in agreement with those introduced in literature.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/631
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

REACTOR CONTROL SYSTEMS WITH MOVING DETECTORS AND THEIR FURTHER PERSPECTIVES

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Since the beginning of 1958, the development of nuclear reactor control using servomechanism with moving detectors is carried out in Czechoslovakia. Advantages of these systems are as follows:

(a) wide working range
(b) universal applicability
(c) availability for complex automatization of control with simple means.

The principle of function of systems with moving detectors is described for information.

After a principal try-out of these systems and after elaboration of their theory, properties of systems with moving ionization chambers were examined using an experimental installation, that was a prototype of full automatic control system of a reactor. Further, characteristics of servosystems with moving detectors of pulse type were investigated. On the basis of satisfactory results of all investigations, a full automatic control system with moving fission chambers was designed and put into operation during the year 1961 on Czechoslovak research reactor VVR-S. Since the time the system works in normal operation of the reactor up to this time. The device carries out the full automatic check-up on the self-readiness, the start of the reactor to a preset level, power control under both steady and transient states, the continuous check-up on self-correctness and on state of the reactor. Results of its operation confirm expected advantages of systems with moving detectors.

All principal questions for using systems with moving detectors on most of contemporary research reactors are solved now. Technical improvement of their elements must be made. Other problems arise by using these systems on power reactors, where special arrangements must be executed to ensure suitable distribution of neutron flux along the detector channels. Analyses show that advantages of systems with moving detectors are a sufficient reason for additional measures which may ensure physical conditions of operation of these systems of these reactor types, where these conditions are not a natural property of medicine surrounding the reactor core.

In Czechoslovakia now an automatic control system for a heavy water zero power reactor is designed using this principle. The reactor of Czechoslovak first nuclear power station also will be provided with an experimental device of this type in order to obtain an experience of operation in conditions of a large power reactor. In both these cases using of the digital technique is supposed.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/632
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

SOME ITERATIVE METHODS FOR THE CONSTRUCTION OF SOLUTIONS OF NEUTRON TRANSPORT EQUATIONS

Ivo Marek

The neutron transport equations can be written in the time independent case in the form

$$Lx = Bx + \lambda Cx + s, \tag{1}$$

where the operators L, B, C and the vectors x, s have the following meaning: The operator L denotes the diffusion and the absorption, B - the scattering, C - the fission; the vector x denotes the density of neutrons or the neutron-flux, s denotes the exterior sources. Let the equation (1) describe the neutron transport in a given bounded domain $\mathcal{M}$. The form of the operators L, B, C depends on the approximation in which the neutron transport is concerned (see G.J. Habetler, M.A. Martino, G. Birkhoff, G.I. Marčuk, I. Marek, R.S. Varga, V.S. Vladimirov). In papers of these authors it is proved, beside other results, that the equation

$$Lx = Bx + \lambda Cx \tag{2}$$

has a solution x_o which is positive in $\mathcal{M}$. This solution corresponds to the positive characteristic value λ_o, for which the inequality $|\lambda| > \lambda_o$ holds for every different characteristic value λ of the equation (2). Moreover this equation has no non-negative solution in $\mathcal{M}$ other than Cx_o, where C is a positive constant. This fact is a consequence of properties of the operators $(L - B)^{-1}C$, $C(L - B)^{-1}$. Easily it can be proved that the value $\mu_o = \lambda_o^{-1}$ is the dominant eigenvalue of the operator T, where T denotes one of the operators $(L - B)^{-1}C$, $C(L - B)^{-1}$.

To construct the eigenelements of the equation (2) and the solutions of the equation (1) there are suitable the iterative methods. Our purpose is to present a general iterative schedule and to prove its convergence.

Let X be a Banach space of vector-functions on $\mathcal{U}$. Let y'_n, z'_n, y' be continuous functionals, for which

$$y'_n(x) \longrightarrow y'(x), \quad z'_n(x) \longrightarrow y'(x)$$

for every $x \in X$ and

$$y'_n(\lambda x) = \lambda y'_n(x), \quad z'_n(\lambda x) = \lambda z'_n(x)$$

for $x \in X$ and $\lambda > 0$.

Let the relations $y'_n(x) \geq 0$, $z'_n(x) \geq 0$ hold for every non-negative vector $x \in X$. Thus $y'(x) \geq 0$ hold for such vectors. Further let $y'(x_o) > 0$, where x_o is the positive eigenvector of the equation (2).

Let us put

$$x_{(n+1)} = \mu_n^{-1} T x_{(n)},$$

where $x_{(o)}$ is a positive vector. Further let be

$$\mu_n = \frac{z'_n(Tx_{(n)})}{y'_n(x_{(n)})}$$

Theorem 1. The sequences $\{\mu_n\}, \{x_{(n)}\}$ are convergent:

$$\mu_n \longrightarrow \lambda_o^{-1}, \quad x_{(n)} \longrightarrow c x_o,$$

where $Lx_o = Bx_o + \lambda_o C x_o$ and c is a positive constant.

To construct solutions of the equation (1) can be used the following formulae:

$$z_{n+1} = \frac{\lambda_{(n)}}{\lambda_{(n)} - \lambda} w_{n+1} + \sum_{n=o}^{n} w_k ,$$

where $\lambda_{(n)} = \mu_n^{-1}$, $w_n = (\lambda T)^k w_o$ and $w_o = (L - B)^{-1} s$ if $T = (L - B)^{-1} C$ and $w_o = C(L - B)^{-1}$ if $T = C(L - B)^{-1}$.

Theorem 2. Let the inequality $|\lambda| < \lambda_o$ be satisfied. Then there exists one and only one solution z of the equation (1), where s is an arbitrary element in X. Moreover in the norm of X it holds $z_{n+1} \longrightarrow z$. The rapidity of the convergence is greater than in the usual method of successive approximations.

Using special functionals y'_n, z'_n we obtain on the one hand some known iterative methods, on the other hand till now not used iterative processes.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/633
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

OPTIMUM DIGITAL INFORMATION HANDLING AND NUCLEAR ENGINEERING INSTRUMENTATION

Pavel Kovanic

The signals treated by nuclear physics and engineering instrumentation are subjected in many cases to statistical disturbances the presence of which is their characteristic property. These disturbances are due to the statistical nature of physical processes, information about which must be given by means of treatment of the input signals. The problem of the optimum treatment of information in the presence of statistical disturbations is solved theoretically by the theory of Wiener, which is directly applicable for the instrumentation having a continuous physical quantity at their inputs and outputs. At high requirements on accuracy, the applicability of this type of apparatus and filter is limited. In addition, it is not always simple to design the instrumentation which approximates the characteristics of an optimum mathematical model according to Wiener theory using the real details.

It is necessary, therefore, to pay attention to the use of digital computers. The following important problems arise, namely:

(1) What algorithm of the digital computer is to be used?
(2) How will the requirements on the accuracy of the results and on the speed of obtaining them affect the requirements on complexity and calculation speed of the computer?
(3) What is the theoretical limit of accuracy regarding the given problem and how do the results computed by the computer approximate to the theoretical limit at the selected algorithm and at selected computer parameters?

Interesting results, i.e. the relatively simple digital computer structure, the possibility to determine simply the theoretical accuracy limit and at the same time the possibility of a "nearly complete" exploitation of the input information, were obtained using following algorithm: the input signal is a

vector of a finite dimension. The components of this vector are given by the finite series of sums of non-random and random function values, the basic statistical parameters of the latter being given. The output signal is obtained as the scalar product of the input vector and the vector-operator. The components of the vector-operator are determined by the requirement of the given analytical accuracy, by the minimum result dispersion and by the type of the linear transformation applied to the input non-random function. Special cases of this problem are as follows: digital interpolation, extrapolation and filtration of the input values of the non-random function, its derivatives, integrals and/or its linear transformations.

These operations are carried out on the base of inaccurate values of the input function, which are given at different points of the finite observation interval. The digital computer has the same task as the optimum filters with a finite memory, which are fed with continuous signals.

The merit of the selected algorithm is the possibility to compute the vector-operator for the given task once and for all and to simplify the digital computer in this way. The passage to another problem is enabled by the digit (vector-operator components) intercharge without any structure and instrumentation component change.

Fundamental theoretical results necessary for practical computations are mentioned in the report. The examples of using this approximation to some nuclear physics and engineering problems are given. Accurate reactor power and period measurements, reactor prompt reactivity measurement using reactor kinetics equations etc. are given as possible utilization regions. The necessity of a passage to digital computers increases with increasing the accuracy requirements and with increasing the transformation complexity which is to be performed by the computer being especially affected by the statistical components of the input signal.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/634
NETHERLANDS
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

PREPARATION OF UO_2 AND ThO_2-POWDERS IN THE SUBSIEVE RANGE

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In the course of the aqueous homogeneous suspension reactor program the need for UO_2 and ThO_2 powders with specific properties arose. The specifications that were to be met included: mean diameters varying from 4 - 10 μ, small standard deviation, low specific surface area, smooth and preferably spherical. Moreover the process should be of the once-through type (no recycle).

Three processes were developed that could meet the above mentioned requirements. It was shown, however, that the particle size could easily be varied, with the result that the processes can also be adapted to other requirements. Moreover it would seem that the processes are well suited to remote control and thus might be preferably used for the refabrication of fuel after reprocessing.

1. The process for ThO_2 or ThO_2-UO_2 mixed oxide is based on the emulsion-gelation ("sol-gel") method. According to this process a ThO_2-sol is dispersed in an organic liquid and the resulting droplets are subsequently gelled. After dehydration and filtration the particles are fired to obtain the low surface area powder. Although primarily developed to produce fuel for the suspension reactor, the process also proved to be applicable to the production of fuel particles with a size up to 1 mm.

2. The first process for the production of UO_2 is based on the precipitation of a basic uranyl-sulphate from a homogeneous

solution. Use is made of the slow hydrolysis of a partly neutralized uranyl-sulphate solution. After reduction and sintering of the precipitate a powder with a very narrow size distribution is obtained.

3. In the second method for the production of UO_2 urea delivers the precipitating NH_3 to a uranylnitrate solution.
With this process at least 8 different types of precipitate can be prepared. The chemistry of this urea process will be discussed in some detail.

One of the modifications, an amorphous one, was produced in a narrow size-range around 10 μ. The mean particle size can be varied by changing the rate of stirring during precipitation. Products with particle diameters up to 30 μ have been obtained.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/635
NETHERLANDS
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

THE AQUEOUS HOMOGENEOUS SUSPENSION REACTOR PROJECT

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This paper gives a survey of the present status of the development of the Aqueous Homogeneous Suspension Reactor Project. This project aims at investigating the application of aqueous oxide suspensions as reactor fuel.

Some features of the mean experimental facility, the Kema Suspension Test Reactor (KSTR) which is under construction, are presented. The design of this test reactor is based on results of the research and development program carried out during the last few years. Typical results in connection with the application of aqueous suspension fuels are reported.

The experimental work has confirmed the important role of colloid chemical phenomena in the behaviour of suspensions. Since this requires colloid chemical investigations at elevated temperatures (up to 250°C) a new method has been developed for this purpose. Results obtained with this method are given.

The predominant influence of the colloid chemical conditions was clearly revealed by the results obtained with a low temperature zero-energy experiment.

Thus it appeared that the reactor behaviour, in particular the reactor noise, depends strongly on these conditions.

Measurements on the range of fission products in UO_2 were carried out in the zero-energy assembly, leading to figures of

4.5 - 8 μ for the various isotopes. The retention of fission fragments in the fuel particles proved to be higher for a flocculating suspension than for a completely dispersed one.

Activation methods have been used to determine the thermal and epithermal neutron flux distribution inside the core vessel.

It was demonstrated that with a circulating suspension very stable and safe reactor operating conditions can be achieved as well as a time independent distribution of the suspension particles in the reactor vessel together with a constant overall concentration. This holds both for colloid chemically stable and unstable conditions.

Irradiation experiments with fuel suspensions in capsules again emphasized the importance of the colloid chemical properties. In several irradiations no fragmentation was observed unless the system was colloid chemically stable. A tentative explanation of the observed phenomena is presented.

Finally some technological aspects of the KSTR are discussed:

Hydrodynamic considerations and measurements leading to the present design of the core vessel together with the time residence distribution in this vessel.

Erosion phenomena caused by fast or irregularly flowing suspensions.

The applications of specially developed hydrocyclones for the separation of gases or fines from the suspension liquid.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/636
NETHERLANDS
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

THE DEVELOPMENT OF NUCLEAR ENERGY IN RELATION TO THE RESERVES OF NUCLEAR MATERIALS

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Many detailed studies have been made about the cost price of nuclear energy. Particularly the fuel cycle costs often have been discussed in detail, However, the influence of the cost price of the nuclear materials, either natural uranium or thorium as a function of the amount available for that price is seldom taken into consideration. Furthermore, although the necessity of breeder reactors for a future large scale nuclear energy production is often stressed, the influence of the cost price of the first nuclear charge is frequently neglected.

In this paper the material requirements for start up and operation of a future large scale nuclear program of nuclear power plants for electricity production are discussed, using the following assumptions:

1. From 1970 - 2000 about 50% of the newly installed electric production capacity will be nuclear;
2. The doubling time of the installed capacity will be 12 years in the period 1962 - 2000;
3. The load factor of the nuclear power stations will be 75%.

Using raw nuclear material reserves from estimates of the U.S.A.E.C. and Euratom, the investment and consumption of nuclear materials is discussed for present day power reactors, high conversion reactors and fast or thermal breeder reactors.

The following results obtained need to determine the

future development and application of nuclear energy:

1. The consumption of nuclear materials in present day power reactors, either enriched or not, is so inefficient that with these reactors the presently known supply of natural uranium at prices up to 5 times the current price will be wasted by 1983.
2. The expenditure of uranium and thorium in breeder reactors is so efficient, that even very expensive nuclear materials abundantly available, can be used.
3. If Pu is produced in present day power reactors as a start up charge for breeders, two or three times more natural uranium will be required for this Pu-production, than by using U-235 which is produced in a diffusion plant as a start up charge.
4. For the first start up charge of breeder reactors the use of rather cheap natural uranium as the primary material, will be required to prevent an excessive capital investment.
5. In fast breeder reactors the start up charge of fissile materials is two or three times higher than in thermal homogeneous breeder reactors.
6. In high conversion heavy water reactors at least the nuclear equivalent amount of Pu can be produced as the amount of U-235 to be extracted from the same quantity of natural uranium.

The conclusions from these results are:

1. Prospection for cheap uranium resources is urgent;
2. The development of breeder reactors is essential for the future of nuclear energy;
3. It is undesirable to use present day power reactors in large numbers;
4. High conversion heavy water reactors can be accepted for an intermediate period, if breeder reactors are not yet sufficiently developed.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/637
NETHERLANDS
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

The Separative Power of Short Countercurrent Centrifuges

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In general countercurrent centrifuges are operated as closed units, which are fed somewhere in the middle of the centrifuge, while at both ends the separated fractions are withdrawn. These centrifuges could be considered as the combination of a rectifying part and a stripping part. In a recent publication[1] it was shown that a symmetrically operated countercurrent centrifuge has an efficiency which amounts to 80% of the maximum separative work that can be performed. It was shown that the decrease of the separative power is due to the fact, that no optimal radial concentration gradient can be maintained along the length of the centrifuge. At both ends the radial concentration gradient approaches the equilibrium concentration gradient, because the net desired material transport approaches zero.

In this work the influence of the feed on the separative power will be considered. In most experiments the feed has to be introduced in the forward stream of the countercurrent, thus giving rise to an asymmetry of the column parameters of the rectifying and the stripping part. This asymmetry can be corrected for only partially, by shifting the feedpoint upstream.

It is shown that a further reduction of about 15% of the separative power results from this asymmetry.

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1) Los, J., Z.f. Natf. To be published Jan. 1964.